



LAWRENCE  
LIVERMORE  
NATIONAL  
LABORATORY

# Graphical User Interface Software for Gross Defect Detection at the Atucha-I Plant

A. C. Wong, S. Sitaraman, Y. S. Ham, O. Peixoto

May 14, 2012

Institute of Nuclear Materials Management 2012  
Orlando, FL, United States  
July 15, 2012 through July 19, 2012

## **Disclaimer**

---

This document was prepared as an account of work sponsored by an agency of the United States government. Neither the United States government nor Lawrence Livermore National Security, LLC, nor any of their employees makes any warranty, expressed or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States government or Lawrence Livermore National Security, LLC. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States government or Lawrence Livermore National Security, LLC, and shall not be used for advertising or product endorsement purposes.

# Graphical User Interface Software for Gross Defect Detection at the Atucha-I Plant

Alden Wong<sup>1</sup>, Shivakumar Sitaraman<sup>1</sup>, Young S. Ham<sup>1</sup>  
and  
Orpet J.M. Peixoto<sup>2</sup>

<sup>1</sup> Lawrence Livermore National Laboratory, 7000 East Avenue, Livermore, CA 94550

<sup>2</sup> Brazilian-Argentine Agency for Accounting and Control of Nuclear Materials,  
Rio de Janeiro, Brazil

## ABSTRACT

At the Atucha-I pressurized heavy water reactor in Argentina, fuel assemblies in the spent fuel pools are stored by suspending them in two vertically stacked layers. This introduces the unique problem of verifying the presence of fuel in either layer without physically moving the fuel assemblies. Movement of fuel, especially from the lower layer, would involve a major effort on the part of the operator. Given that the facility uses both natural uranium and slightly enriched uranium at 0.85 w% <sup>235</sup>U, and has been in operation since 1974, a wide range of burnups and cooling times can exist in any given pool. Additionally, while fuel assemblies are grouped together in a uniform fashion, the packing density from group to group can vary within a single pool. A tool called the Spent Fuel Neutron Counter (SFNC) was developed and successfully tested at the site to verify, in an *in-situ* condition, the presence of fuel up to burnups of 8,000 MWd/t. Since the neutron source term becomes a nonlinear function of burnup beyond this burnup, a new algorithm was developed to predict expected response from the SFNC at measurement locations covering the entire range of burnups, cooling times, and initial enrichments. With the aid of a static database of parameters including intrinsic sources and energy group-wise detector response functions, as well as explicit spent fuel information including burnups, cooling times, enrichment types, and spacing between fuel assemblies, an expected response for any given location can be calculated by summing the contributions from the relevant neighboring fuel assemblies. Thus, the new algorithm maps the expected responses across the various pools providing inspectors with a visual aid in verifying the presence of the spent fuel assemblies. This algorithm has been fully integrated into a standalone application built in LabVIEW. The GUI uses a step-by-step approach to allow the end-user to first calibrate the predicted database against a set of measurements with SFNC at selected locations where spent fuel is present. Once the database is calibrated it can be used to detect gross defects by comparing the measured signal to the one predicted by the database with differences beyond a set tolerance indicating missing fuel.

## INTRODUCTION

In the area of nuclear safeguards, the process of verifying the presence of spent fuel assemblies (SFAs) can prove challenging. This is particularly the case with Atucha-I, a pressurized heavy water reactor, where SFAs are stored in pools with two vertical layers. The assemblies are suspended from hangers (*perchas*) attached to large beams. Each pool has multiple beams with hangers from which SFAs can be suspended either on one side or on both sides of each beam. The Digital Cerenkov Viewing Device (DCVD) is not an effective tool even for the top layer since a large number of SFAs are in the low burnup range and have cooling times of 20 years. This poses a unique problem of verifying the presence of fuel assemblies in the pool. Movement of the fuel is virtually impossible without removing large numbers of SFAs from their stored positions. This would be particularly the case for the lower layer of SFAs which are also partially hidden from view looking down into the pool. It is therefore imperative to develop means of performing this verification in an *in-situ* condition.

Figures 1 through Figure 3 show the basic configurations. Figure 1 shows the top view of the configuration where assemblies are suspended from only one side. The alternate configuration of suspending assemblies from both sides is shown in Figure 2. Lastly, Figure 3 shows the axial view of the pool with the two layers of vertically stacked assemblies.

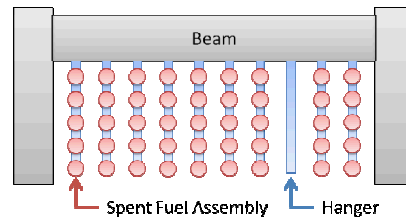


Figure 1: Top View of Single Side Hanger for Storing Fuel Assemblies

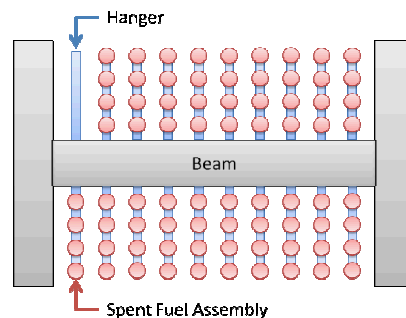


Figure 2: Top View of Dual Side Hanger for Storing Fuel Assemblies

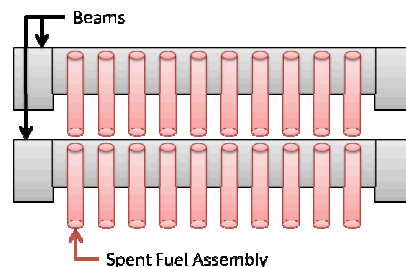


Figure 3: Side View of Hanger for Storing Fuel Assemblies

Additionally, for both the single side and dual side configurations, there is the option of having low density or high density packing. Specifically, a single side low density configuration will suspend five assemblies per hanger while a high density format will suspend six assemblies per hanger. In the case of the dual side configuration, low density packing will suspend four assemblies per side per hanger, while high density packing will suspend five assemblies per side per hanger. It is important to note that the lower level of the pool does not necessarily copy the same hanger structure as the upper level. A dual side low density configuration can exist underneath a single side low density configuration. Aside from the special considerations of the physical arrangement of SFAs in the pools, the SFAs have a wide range of burnups (5000-11000 MWd/t) and cooling times (2-40 years). The initial fuel composition consisted of both natural uranium fuel as well slightly enriched uranium at 0.85 w%<sup>235</sup>U. Thus, the underlying algorithm to predict and map the neutron signals at various locations in the pools must account for all of these factors.

A detection system called the Spent Fuel Neutron Counter (SFNC) [1, 2] was previously developed to allow a fission chamber to be inserted into the space among a cluster of four adjacent spent fuel assemblies (see Figure 4). This enabled neutron signals to be measured in both layers in an in-situ condition. The algorithm used in this system assumed linearity between the neutron source term and the

burnup. This assumption is valid up to burnups of around 8000 MWd/t. However, owing to the increase in initial enrichment, the discharge burnups of spent fuel can be as high as 11,000 MWd/t. In this range of burnup, the assumption of linearity with the source breaks down. As a result of this, a new algorithm was developed to predict and map the expected signals at any location in a spent fuel pool. The following sections describe an algorithm that was developed to estimate the response of the SFNC and how gross defects can be determined by comparing the ratio of estimated response and the measured response to a specified threshold value. The GUI design and functionality, with actual screenshots, is then described. Finally, results using previously collected data are presented to validate the software's capabilities

### ALGORITHM FOR ESTIMATING NEUTRON SIGNAL

The general idea behind the algorithm is to sum the expected contributions to the neutron count rate by nearby spent fuel assemblies. This is easily understood by the diagram presented in Figure 4. Given that the center spot is the measurement point, the estimated neutron count rate can be determined by adding all the contributions from each of the 36 nearby spent fuel assemblies. Assuming the horizontal and vertical spacing between assemblies is the same (Figure 4 – Left), each letter represents a unique (x,y) distance away from the center. In the case of asymmetric horizontal and vertical spacing (Figure 4 – Right), there will be slight difference in the contributions from B and BB even if they are the exact same burnup and cooling time, due to their differing (x,y) distance from the measurement point. Likewise, D will differ from DD and E will differ from EE. The 6x6 configuration is more than adequate to capture contributions from neighboring SFAs to the detector signal.

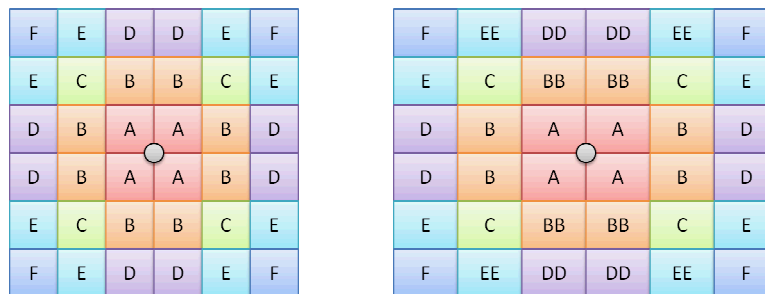


Figure 4: 6x6 Grid of Nearby Assemblies (Left – Symmetric Spacing; Right – Asymmetric Spacing)

Earlier studies dealt with a methodology based on developing adjoint fluxes calculated using a hybrid stochastic-deterministic approach that can be used in conjunction with enrichment, burnup, and cooling time based intrinsic source spectra to calculate the expected signal at the detector location [3, 4, 5, 6]. However, since these datasets were constructed, the pool configurations at Atucha-I were changed. The adjoint fluxes are dependent on the distance between the sources and the detector. Since regenerating these fluxes together with the associated fission matrix coefficients would be more time consuming, a more straightforward, purely stochastic method was used to build source-to-detector transfer functions. These energy dependent transfer functions for the different inter-assembly pitches were developed using the Monte Carlo radiation transport code, MCNP [7]. The transfer functions combined with appropriate SFA source spectra form the basis of the algorithm used in the GUI software to map predicted neutron signals.

Radiation transport calculations were performed for each intra-assembly pitch present in the various Atucha spent fuel pools. The basic methodology consisted of starting sources in each of the assemblies (e.g. A, B, BB, etc., as shown in Figure 4), and tallying the contribution to the detector by starting a single neutron in each of the 47 energy groups used. These transfer functions in each group can be multiplied by the actual intrinsic source strength in that group. The total over each of the 47 groups for all 36 assemblies can then be calculated and the grand total over all the groups would give the expected signal for that detector location. The calculations were performed with isotopics based on an average burnup of 8500 MWd/t and a cooling time of 20 years. The transfer functions are relatively insensitive to the isotopics

used as described in the previous studies based on adjoint fluxes [5]. The intrinsic source spectra were evaluated using ORIGEN-ARP [8] and an extensive set of source spectra was calculated for the different burnups and cooling times at each of the two initial enrichments [5].

In order to calculate the estimated total neutron count rate, two sets of fixed data are required. A table of intrinsic source terms for each of the 47 energy groups is given for intervals of 1000MWd/t burnups and 10 selected cooling times between 0.03 years to 40 years. The range of burnups is 4000-12000 for natural uranium and 9000-20000 for slightly enriched uranium. Linear interpolation is used to obtain the correct source spectrum at a specific burnup and cooling time. The second set of data is a table of transfer functions which weights the contributions of each of the 47 energy groups depending on the spent fuel assembly's distance from the point of measurement.

Details of the SFAs in the spent fuel pools were obtained from the operator of the plant and included the locations of all the spent fuel assemblies, their corresponding enrichments, burnups, and cooling times, and the spacing between the assemblies. The software implements the following steps to generate the predicted responses of the SFNC:

For a given measurement location,

1. Pick a spent fuel assembly within the 6x6 grid of nearby assemblies
2. Determine the spent fuel assembly's characteristics:
  - Enrichment (NU or SEU)
  - Burnup
  - Cooling time
3. Calculate the intrinsic source term for this specific spent fuel assembly by linearly interpolating burnup and cooling time from the intrinsic source terms table
4. Based on the location of the spent fuel assembly determine which values to use from the table of transfer functions
5. For each energy group, multiply the intrinsic source term value from step 3 by the transfer function value from step 4 and sum all resulting values. This represents the contribution to the neutron signal at the detector from this assembly
6. Add neutron signal to a running total
7. Repeat until all spent fuel assemblies in the 6x6 grid have been analyzed

Because the SFNC's detector efficiency is unknown, the estimated and measured values will not be equivalent in absolute terms. However, the results can be scaled by a calibration factor and then be subsequently used to detect possible gross defects. This is described in the following section.

### **CALIBRATION FACTOR AND THRESHOLD VALUE FOR GROSS DEFECT DETECTION**

To convert the estimated neutron count rate numbers to more meaningful numbers, they are normalized by a calibration factor ( $E$ ). Specifically, for spent fuel assemblies which can be verified by other means,  $n$  number of measurements ( $x_i$ ) can be taken with the SFNC and the corresponding estimated neutron count rate values ( $r_i$ ) can be calculated as well. Assuming the detector efficiency and other sources of error can be rolled into a constant calibration factor, this calibration factor can be calculated by minimizing the square error function described by the equation on the left. This is done by taking the derivative of the function with respect to the calibration factor and setting it equal to 0 as seen in the equation on the right:

$$Y = \sum_i^n (x_i - E * r_i)^2 \rightarrow \frac{d}{dE} \sum_i^n (x_i - E * r_i)^2 = 0$$

The resulting equation defines the calibration factor as the ratio of two sums. The dividend is the sum of the products between the measurement values and the corresponding estimated neutron count rate values. The divisor is the sum of the squares of the estimated values. This is shown in the following equation:

$$E = \frac{\sum_i^n (x_i * r_i)}{\sum_i^n (r_i^2)}$$

With the calibration factor calculated, measurements can be taken using the SFNC in locations where the presence of spent fuel assemblies needs to be verified. The estimated neutron count rate values should also be calculated for the corresponding locations. The calibration is then applied to all of the estimated values. Finally, these normalized estimated values can be compared to the measured values and the percent deviation can be calculated by the following equation:

$$Deviation = \frac{|r_i * E - x_i|}{r_i * E}$$

Because the calibration factor minimizes the square error between the estimated neutron count rate and the measured value from the SFNC, even applying the calibration factor on the original set of data will likely not produce exact matches. Therefore, a non-zero threshold must be defined to determine whether or not a gross defect exists. Under typical conditions, a threshold between 10% and 15% is ideal.

## GUI SOFTWARE

The software, which was designed in LabVIEW [9], was created with the idea of simplicity in mind. A step-by-step approach with a very clear and visual interface attempts to minimize user-error. The main screen is shown below in Figure 5. The left side presents the user with the current progress. The bottom bar allows the user to restart, read the help screen, or quit, at any time.

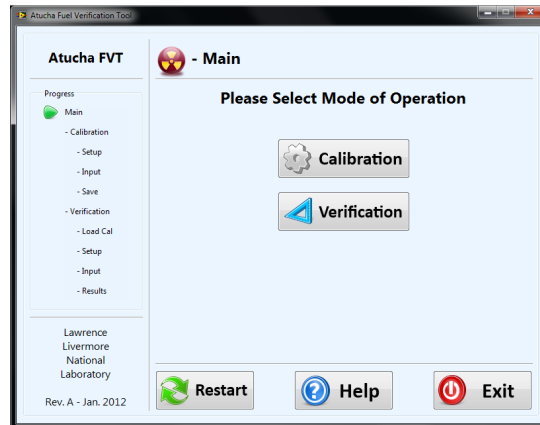


Figure 5: Main Screen of the Software

Figure 6 is an example screen of the user choosing the pool and number of calibration points they wish to use. When possible, everything is done with simple drop-down menu options or buttons. Previous and Next buttons control the flow of the process. Next buttons are disabled until all required information is validly entered. In Figure 7, the user inputs the locations being measured by clicking on a cell in a grid which represents the pool. The user will also enter the measurement value obtained from the SFNC for the corresponding location; this is the only time the user is required to enter values via keyboard. The second screen capture in Figure 7 shows how corresponding pool maps color coded to initial enrichments, burnups, or cooling times, can be viewed to provide more detailed information to the user.

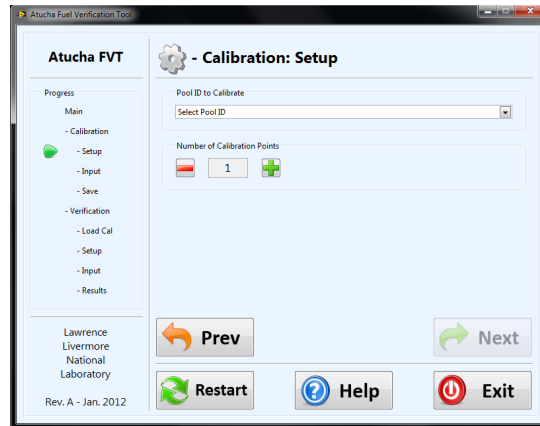


Figure 6: The User Interacts Via Buttons and Drop-Down Menus

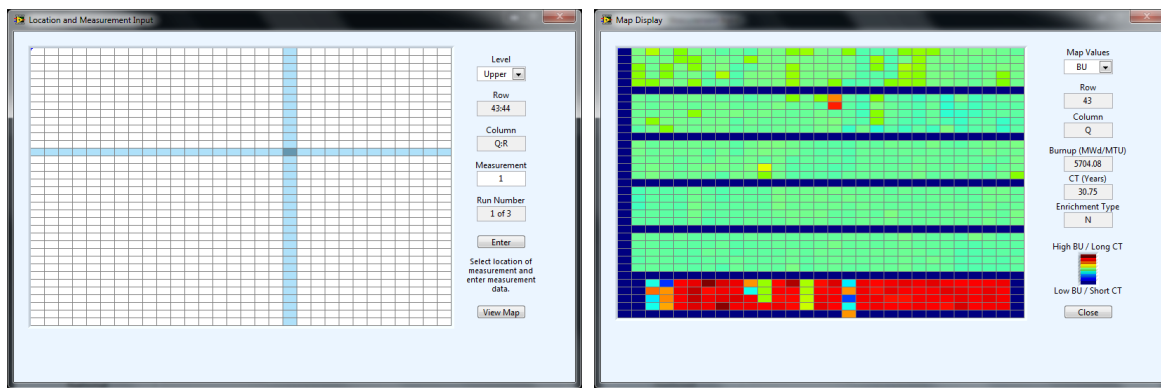


Figure 7: The User Chooses a Location (Left); Corresponding Burnup Color Coded Pool Map (Right)

The two screen captures in Figure 8 demonstrate the calibration factor being calculated and the calibration file being saved. Similarly looking screens are presented for loading the calibration file during the verification phase and generating the full report after the verification phase has been completed.

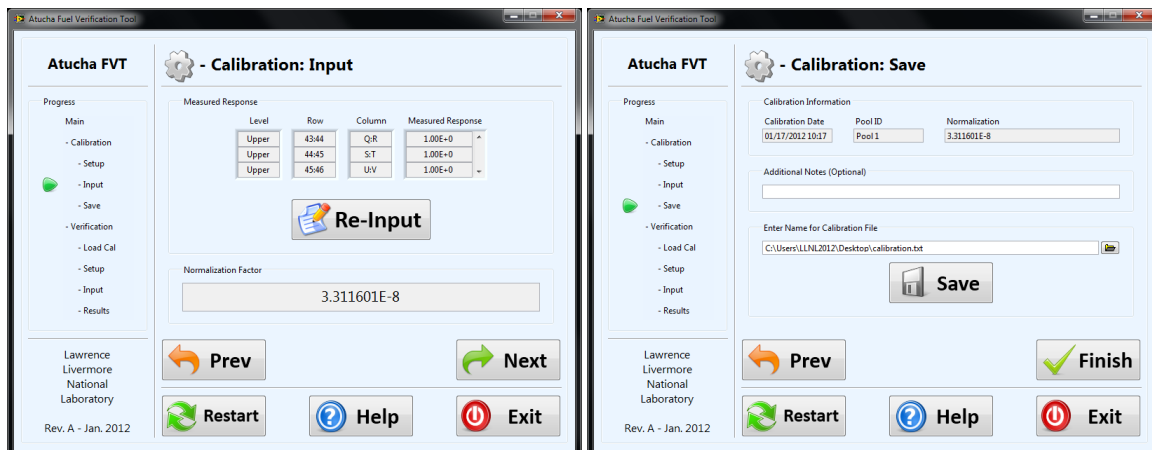


Figure 8: Calibration Factor Being Calculated (Left); Calibration File Being Saved (Right)

Finally, Figure 9 depicts the screen shots in which the user is clearly notified that all measurements were within the specified tolerance or that one or more measurements lies out of the tolerance limit.



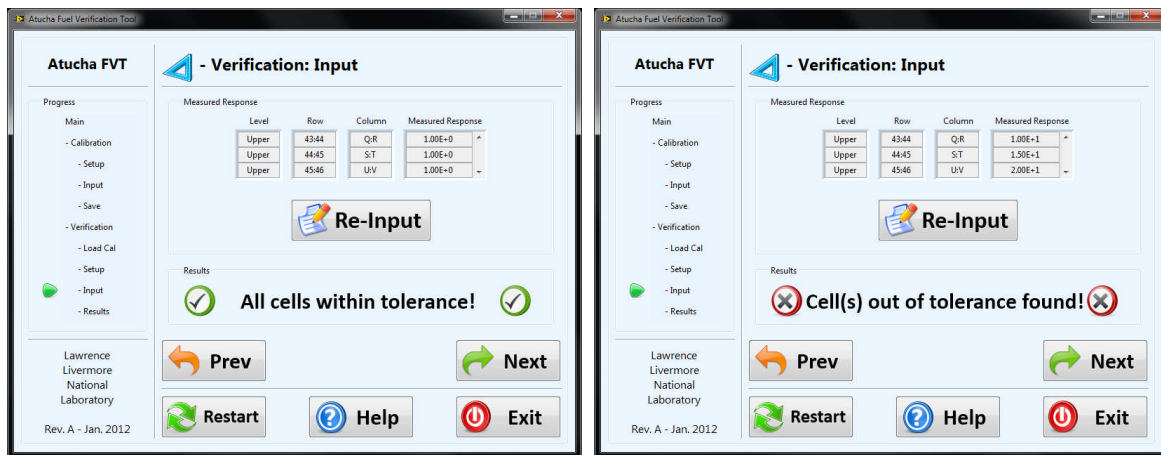


Figure 9: Measurements Within Tolerance (Left); Measurements Out of Tolerance (Right)

### INITIAL VALIDATION OF THE SOFTWARE

Measurements were made in 2002 at the Atucha-I facility with the SFNC tool. Of all the data taken, there were 25 which were used in the GUI software for validation purposes. These 25 measurement points were those in which four spent fuel assembly locations surrounded the SFNC (see “standard” locations in Figure 10). There were some other data points where only two spent fuel assembly locations surrounded the SFNC (i.e. measurement locations on the boundaries of each SFA grouping defined by the beam— see “edge” locations in Figure 10). The reason is that between two edge locations, there is a non-standard gap that currently is not modeled by the GUI software pending more information from the operator as to the extent of this gap.

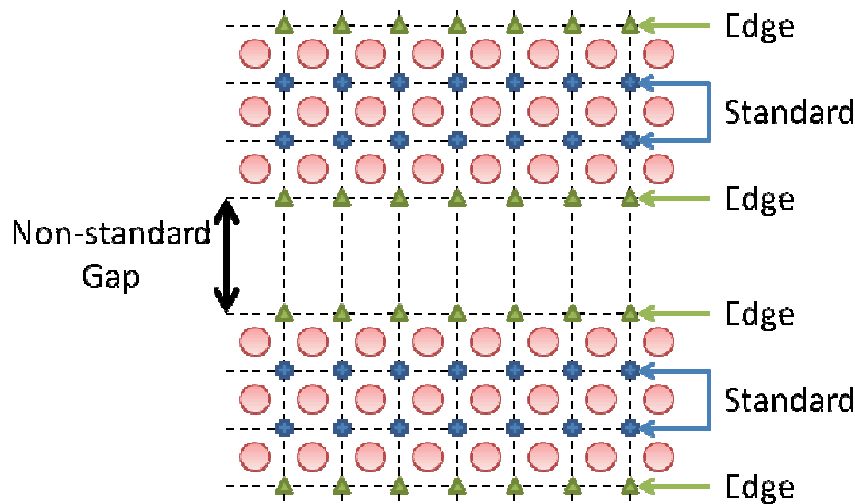


Figure 10: Mock Pool Mapping Describing “Edge” vs “Standard” Measurement Locations

For the purposes of this validation, 25 calibration factors were determined for each of the 25 measurements through the calibration mode of the GUI software (see Table 1). These 25 calibration factors were arithmetically averaged to a single calibration factor of  $5.79\text{e-}7$ . This calibration factor was then used in the verification mode of the GUI software. In the ideal case, the ratio of the theoretical values and measured values is the same constant and the average calibration factor is thus equal to all 25 individual calibration factors. If this were true, the verification mode would indicate that all measurement locations would match the respective calibrated theoretical signal. However, because the ratio of theoretical values and measured values do differ slightly, the overall calibration factor represents the scaling needed to minimize the overall square error. Table 1 presents the percent deviation of the

measured signal from those calculated during this validation process. It is clear that all points (except #25) remain under 15% and that 19 out of the 25 are under 10%.

Measurement	Calibration Factor	Percent Deviation
1	6.41E-07	10.80%
2	5.35E-07	7.50%
3	6.44E-07	11.30%
4	6.06E-07	4.80%
5	5.96E-07	3.10%
6	6.14E-07	6.10%
7	6.25E-07	8.10%
8	6.39E-07	10.40%
9	6.36E-07	10.00%
10	6.12E-07	5.80%
11	6.24E-07	7.90%
12	5.64E-07	2.50%
13	6.24E-07	7.90%
14	5.39E-07	6.80%
15	5.87E-07	1.40%
16	5.36E-07	7.30%
17	5.57E-07	3.60%
18	5.38E-07	6.90%
19	5.45E-07	5.70%
20	5.08E-07	12.10%
21	5.65E-07	2.30%
22	5.44E-07	5.90%
23	5.35E-07	7.40%
24	5.92E-07	2.30%
25	4.63E-07	19.90%

Table 1: Calibration Factors

## CONCLUSIONS

The overall design of the GUI was intended to provide a simple aid in the verification process at the Atucha-I facility. It is relatively easy to use via large buttons and intuitive data entry methods. Calculations are completed within a few seconds, allowing the user to quickly obtain the results. However, manual data entry leaves the door open for user-created errors in both the calibration and verification process. Future designs would likely benefit from a method of interfacing from the SFNC tool directly and automatically extracting the results. Lastly, because the algorithm is founded on certain underlying assumptions as discussed earlier, a certain tolerance limit, typically in the range 10-15%, must be applied when comparing predictions to measurements. The initial validation of the methodology and GUI software show encouraging trends that a map of expected neutron signals can be developed for each pool within this tolerance range. Thus the SFNC coupled with this enhanced software provides a repeatable method of determining gross defects in both layers of SFAs in the various spent fuel pools located at the site.

Further validation of the software will be performed with more measured data that is in the process of being obtained. A field demonstration of the system is also in the planning stages and is likely to be conducted later in 2012. The enhanced SFNC will help close a deficiency in the safeguards portfolio of ABACC that has existed since the Atucha-I plant started discharging fuel at higher burnups.

## ACKNOWLEDGEMENT

This work was performed under the auspices of the U.S. Department of Energy by Lawrence Livermore National Laboratory under Contract DE-AC52-07NA27344.

The authors are grateful to Mr. W. Walters and Prof. A. Haghighat of Virginia Tech for the referenced earlier work that generated the set of source terms used and the sensitivity studies performed to identify parameters of importance that affect the detector responses.

The authors are also grateful to ARN, Buenos Aires, Argentina for providing information on the spent fuel pools.

## REFERENCES

1. Ham, Y. et al., "Neutron Measurement Techniques for Verification of Closely Packed Spent Fuel Assemblies Stored in a Spent Fuel Pond," *Proceedings of the INMM 44th Annual Meeting*, Phoenix, AZ, July 2003.
2. Peixoto, O.J.M. et al., "A Safeguards Approach Applied to Control the Spent Fuel Assemblies Stored in a Spent Fuel Pond on a Unique On Load Reactor", *Proceedings of the INMM 45th Annual Meeting*, Orlando, FL, July 2004.
3. W. Walters, A. Haghighat, M. Wenner, Y. Ham, S. Sitaraman, "Methodology and Determination of Field of View of Neutron and Gamma Detectors in the Atucha Spent Fuel Storage Pool," *Proceedings of the INMM 50th Annual Meeting*, Tucson, AZ, July 2009.
4. W. Walters, A. Haghighat, M. Wenner, Y. Ham, S. Sitaraman, "Calculation of Sub-critical Multiplication Using a Simplified Fission Matrix Method," *Transactions of the American Nuclear Society Winter Meeting*, Washington, DC, November 2009.
5. W. Walters, A. Haghighat, M. Wenner, S. Sitaraman, Y. Ham, "A Methodology for Determination of Detector Response for Inspection of a Spent Fuel Pool", *PHYSOR 2010-Advances in Reactor Physics to Power the Nuclear Renaissance*, Pittsburgh, PA, May 2010.
6. S. Sitaraman, Y. Ham, W. Walters, A. Haghighat, O. Peixoto, "In-Situ verification of Spent Fuel at the Atucha-I Nuclear Power Plant", *Proceedings of the INMM 52th Annual Meeting*, Palm Desert, CA, July 2011.
7. X5-Monte Carlo Team, "MCNP-A General Monte Carlo N-Particle Transport Code," Version 5.1.40, Los Alamos National Laboratory, February 2006.
8. ORIGEN-ARP, Version 5.1.01, Isotope Generation and Depletion Code, CCC-732, Radiation Safety Information Computational Center, March 2007.
9. NI LabVIEW <<http://www.ni.com/labview/>>